**GEORGIA INSTITUTE OF TECHNOLOGY**
OFFICE OF CONTRACT ADMINISTRATION

# SPONSORED PROJECT INITIATION

Date: January 14, 1981

Project Title: Perform Work in Support of the International Workshop on the Next Major Tokamak Experiment

Project No: E-26-669 (Continuation of E-26-643)

Project Director: Dr. Weston M. Stacey, Jr.

Sponsor: Department of Energy, Oak Ridge Operations; Oak Ridge, Tennessee 37830

Agreement Period: From October 1, 1980 Until ~~September 30, 1981~~ (Contract Period)
(R+D Perf + Rpt Periods) 11/30/81

Type Agreement: Contract No. DE-AS05-79ET52049, Mod No. A004

Amount: $100,000

Reports Required: Quarterly Technical Reports; Progress Report; Final Report; Publication Preprints; Publication Reprints

Sponsor Contact Person (s):

Technical Matters

Dr. C. R. Head
Reactor Systems and Applications Branch
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Office of Fusion Energy
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Washington, D. C. 20545

Contractual Matters
(thru OCA)

Mr. Allen Askew
Research Contracts, Procedures and Reports Branch
Contract Division
Oak Ridge Operations
U.S. DEPARTMENT OF ENERGY
P. O. Box E
Oak Ridge, Tennessee 37830

615/576-0788

Defense Priority Rating: None

Assigned to: Nuclear Engineering (School/Laboratory)

COPIES TO:

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CA-3 (3/76)

GEORGIA INSTITUTE OF TECHNOLOGY OFFICE OF CONTRACT ADMINISTRATION

## SPONSORED PROJECT TERMINATION/CLOSEOUT SHEET

Date 2/10/84

Project No. E-26-669 School/~~Lab~~ Nuclear Engr.

Includes Subproject No.(s)

Project Director(s) Dr. W. M. Stacey, Jr. ~~GTRI~~ / GIT

Sponsor Department of Energy, Oak Ridge, TN

Title Perform Work in Support of the International Workshop on the Next Major Tokamak Experiment

Effective Completion Date: 11/30/81 (Performance) 11/30/81 (Reports)

Grant/Contract Closeout Actions Remaining:

- [x] None
- [ ] Final Invoice or Final Fiscal Report
- [ ] Closing Documents
- [ ] Final Report of Inventions
- [ ] Govt. Property Inventory & Related Certificate
- [ ] Classified Material Certificate
- [ ] Other

Continues Project No. E-26-643 Continued by Project No. E-26-687 (Same Contract DE-AS05-79ET52049)

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Quarterly Progress Report

Period 10/1/80 through 12/31/80

"Perform Work in Support of the International Workshop on the Next Major Tokamak Experiment"

Contract No. DE-AS05-79ET-52049


Weston M. Stacey, Jr.
School of Nuclear Engineering
Georgia Institute of Technology
Atlanta, GA 30332

The U.S.A. participation in the Phase-1 INTOR conceptual design activity was coordinated. This included attending session IV (20-31 October, 1980) of the INTOR Workshop, attending a meeting of the IFRC (17 November 1980) to report on the status of the INTOR Workshop (see attached summary) and direction and review of design work carried out at various institutions within the U.S.A.

With the decisions taken or confirmed at session IV, the preliminary INTOR concept is now adequately defined so that a conceptual design can be completed by June, 1981. This conceptual design will be carried out in sufficient detail to provide a definition of the INTOR concept that is supported by design analyses.

The INTOR objectives and staged operation schedule remain basically the same as those defined in Phase-0, with some clarification of the stage III objectives. INTOR is intended to demonstrate the plasma physics and technological requirements of a DEMO and to serve as an engineering test facility. Demonstration of the plasma physics and of the integration of the technological systems required for a DEMO will be accomplished, for the most part, in stage I during the first three years of operation. The second stage, of about four years duration, will be devoted to engineering testing. The main objective of stage III is to achieve $\sim$5 MW·yr/m² within $\leq$ 10 years after the end of stage II for blanket and other technological systems reliability testing.

The plasma characteristics have evolved slightly from those identified in Phase-0. The INTOR plasma, with minor radius a = 1.2 m, elongation $\kappa = 1.6$, toroidal field at the chamber center of $B_t = 5.5$ and with a current of I = 6.4 MA can produce a thermonuclear power output of P = 620 $MW_t$ and an average neutron wall load of $P_n = 1.3$ MW/m² by operating with $\langle\beta_t\rangle = 5.6\%$. The INTOR plasma can be heated to ignition with 75 MW of 175 keV neutral beam injection power. RF heating is the backup option. A re-evaluation of ignition requirements based upon recent energy confinement data from PLT, PDX and T-10 indicates that the ignition margin of the INTOR plasma is 2.5-3.0. Major physics tasks that have been identified for the remainder of Phase-1 include: clarification of the scrape-off and divertor physics; development of a suitable magnetic field configuration; and development of a burn control scheme.

A single-null poloidal divertor, with the chamber at the bottom, has been selected for impurity control. A double-null poloidal divertor is the backup option. Preliminary physics and engineering analyses indicate that a single-null divertor is feasible, although there is a problem with magnetically forming the inside channel. The divertor will be designed for modular (12) removal and replacement. Major design activities that have been identified for the remainder of Phase-1 include: magnetics for divertor channel formation and separatrix position control during startup; scrape-off region and divertor chamber physics; collector plate design; and engineering design of a replaceable divertor module.

The principal features of the first-wall, blanket and shield have been identified. The structural material will be stainless steel and the coolant will be water. Aluminum is the backup option for the first-wall material. The first-wall will have passively-cooled armor (probably carbon) or equivalent on the inboard part to protect against disruptions. A partial (upper and outboard) tritium-producing blanket will be installed from the beginning to produce $\geq$ 60% of the tritium required for INTOR. The breeding material will be lithium-silicate. A liquid lead-lithium-(bismuth) eutectic is the backup option. Major design activities that have been identified for the remainder of Phase-1 include design of the first-wall and design of the tritium-producing blanket.

A preliminary definition of the engineering testing program and of the resulting requirements upon INTOR design and operation have been achieved. Reactor prototypical blanket testing will be performed in 4-6 test modules each taking up about one square meter of plasma chamber area. Materials and other blanket tests will be performed in smaller channels covering about 4 $m^2$ of plasma chamber surface area. Major activities that have been identified for the remainder of Phase-1 include detailed definition of the engineering test program and scoping design of the experimental components.

A concept and some preliminary designs for a remotely maintainable mechanical configuration have been developed. A semipermanent inboard, upper and lower shield forms the primary vacuum boundary. Twelve blanket sectors fit within this shield. These blanket sectors are partially (upper, outboard) tritium producing blanket and partially (inboard, lower) shield. Once the outboard shield is removed, each blanket sector can be withdrawn horizontally with straightline motion through a window between adjacent toroidal field coils. Blanket test modules are also inserted and withdrawn horizontally through the windows. Major tasks that have been identified for the remainder of Phase-1 include: the development of a self-consistent mechanical configuration; the definition of remote assembly/disassembly procedures: and the analysis of the electromechanical response of the blanket/shield/structural system and the effect upon plasma performance.

Semipermanent, superconducting toroidal and poloidal field coils will be enclosed in a common, semipermanent cryostat. All poloidal field coils external to the toroidal field coils will be superconducting. However, it may be necessary to allow a limited number of internal, resistive poloidal field coils to provide the high voltage pulse needed for plasma breakdown, to form the inner divertor channel, to provide for control of the vertical instability and to reduce the pulsed field loading on the toroidal field coils. The structural support system for the toroidal field coils will make use of coil wedging and/or a bucking cylinder to handle in-plane and centering forces. A shear constraint at the inner leg and a bending constraint of the outer leg will be used to handle out-of-plane forces, and the use of shear panels will be considered if necessary. Major design activities that have been identified for the remainder of Phase-1 include: design of the coils; design of the cryostat; design of a remotely maintainable joint for internal, resistive coils; and design of a structural support system.

Other important design tasks that have been identified for the remainder of Phase-1 include design of a neutral beam injection system, design of tritium fuel cycle and blanket tritium processing systems, design of a pellet fueling system, and design of the shielding system.

The procedure for converging to a single INTOR conceptual design in June, 1981 has been specified. All four countries are developing their national design contributions to the INTOR Workshop in accord with the design specifications and major parameters that have been agreed to at the Workshop sessions. At each Workshop session, further decisions are taken at a greater level of subsystem detail, and these decisions are incorporated in the design contributions of all countries at the next Workshop session. A detailed outline for the final INTOR conceptual design report was developed at session IV. The national contributions to the next two sessions (session V, January 19 - February 6, 1981 and session VI, March 30 - April 10, 1981) will be structured according to this outline in order to facilitate the convergence process. Most design activities will be completed by session VI. A first draft of the INTOR conceptual design report will then be prepared for review, revision and approval at session VII (June 22 - July 3, 1981). A final draft of this report will be available at the end of session VII.

The INTOR Workshop has organized special groups to: insure that the conceptual design meets appropriate safety and environmental criteria; to update the cost/manpower/schedule assessment made during 1979; and to supplement the R&D assessment made during 1979.

GEORGIA INSTITUTE OF TECHNOLOGY

SCHOOL OF NUCLEAR ENGINEERING

Atlanta, Georgia 30332 (404) 894–3720

E-26-669

12 February 1981

Foreign Travel Report

Weston M. Stacey, Jr.
Georgia Institute of Technology
School of Nuclear Engineering
Atlanta, GA 30332
Contract #DE-AS05-79ET-52049

Purpose

Attend Session V of the Phase 1 INTOR Workshop as U.S.A. Steering Committee member.

Results

See attached "Summary of the Phase-1..Session V..INTOR Workshop..."

Attachments: Summary of the Phase-1 INTOR Workshop Upon Completion of Session V

Report of Foreign Travel Cost

Copies to: Dr. C. R. Head OFE/DOE, Washington (1 copy)
Mr. W. A. Mynatt DOE/Oak Ridge Operations (8 copies)
Nancy McHan GIT/OCA
File

4 February 1981

SUMMARY OF THE PHASE - I INTOR WORKSHOP
UPON COMPLETION OF SESSION V, JANUARY 19 - FEBRUARY 4, 1981

With the decisions taken or confirmed at session V, the INTOR concept is now adequately defined so that a conceptual design report can be completed by June, 1981. This conceptual design is being carried out in sufficient detail to provide a definition of the INTOR concept that is supported by design analyses.

The INTOR objectives and staged operation schedule remain basically the same as those defined in Phase-0, with some clarification of the stage III objectives. INTOR is intended to demonstrate the plasma physics and technological requirements of a DEMO and to serve as an engineering test facility. Demonstration of the plasma physics and of the integration of the technological systems required for a DEMO will be accomplished, for the most part, in stage I during the first three years of operation. The second stage, of about four years duration, will be devoted to engineering testing. The main objective of stage III is to achieve $\sim 5$ MW.yr/$m^2$ within $\leq 10$ years after the end of stage II for blanket and other technological systems reliability testing.

The plasma and operational characteristics have evolved slightly from those identified in Phase-0. The INTOR plasma, with minor radius a = 1.2 m, elongation k = 1.6, toroidal field at the chamber center of $B_t$ = 5.5 and with a current of I = 6.4 MA can produce a thermonuclear power output of P = 620 MWt and an average neutron wall load of $P_n$ = 1.3 MW/$m^2$ by operating with $\langle \beta_t \rangle$ = 5.6%. The INTOR plasma can be heated to ignition with 75 MW of 175 keV neutral beam injection power. RF heating is the primary alternative. A re-evaluation of ignition requirements based upon recent energy confinement data from PLT, PDX and T-10 indicates that the predicted alpha-heating power exceeds that predicted for ignition by a factor of about 2. The INTOR plasma will operate with a 200 s burn pulse and a duty factor of $\sim$80%. Burn control will be accomplished by variable toroidal field ripple.

A single-null poloidal divertor, with the chamber at the bottom, has been selected for impurity control. A double-null poloidal divertor is the primary alternative. Physics and engineering analyses indicate that it is possible to magnetically form thd divertor channel and control the separatrix with coils located external to the toroidal field coil. Divertor collector plates wit[h] a predicted lifetime $\gtrsim$ 1 year have been designed. The divertor will be designed for modular removal and replacement.

The principal features of the first-wall, blanket and shield have been identified. The structural material will be stainless steel with a panel type construction, and the coolant will be water. The first wall may last close to the entire lifetime of INTOR, provided that the melt layer which is formed on the inboard wall during disruption is stable. If this melt layer proves to be unstable, the inboard wall will be protected with a passively-cooled graphite armour. Aluminum is the primary alternative structural material.

A partial (upper and outboard) tritium-producing blanket will be installed from the beginning to produce $\gtrsim$ 60% of the tritium required for INTOR. The breeding material will be lithium-silicate or lithium-oxide, the structural material will be stainless steel and the coolant will be water. A liquid lead-lithium-(bismuth) eutectic is the primary alternative breeding material.

A preliminary definition of the engineering testing programme and of the resulting requirements upon INTOR design and operation have been achieved. Reactor prototypical blanket testing will be performed in test modules covering about 8 $m^2$ of plasma chamber area. Materials and other blanket tests will be performed in smaller channels covering about 4 $m^2$ of plasma chamber surface area.

A concept and some preliminary designs for a remotely maintainable mechanical configuration have been developed. A semipermanent inboard, upper and lower shield forms the primary vacuum boundary. Twelve blanket sectors fit within this shield. These blanket sectors are partially (upper, outboard) tritium producing blank[et] and partially (inboard, lower) shield. Once the outboard shield is removed, each blanket sector can be withdrawn horizontally with straightline motion through a window between adjacent toroidal field coils. Blanket test modules are also inserted and withdrawn through the windows.

Semipermanent, superconducting toroidal and poloidal field coils will be enclosed in a common, semipermanent cryostat. All poloidal field coils will be external to the toroidal field coils and will be superconducting, except for a cryoresistive solenoidal OH coil to provide the 100 V pulse for plasma initiation. A passive circuit will be included in the first wall by sector connection for the control of vertical instabilities. The structural support system for the toroidal field coils will make use of coil wedging and/or a bucking cylinder to handle in-plane and centering forces. A shear constraint at the inner leg and a bending constraint of the outer leg will be used to handle out-of-plane forces.

A procedure has been implemented for conveying the national design contributions to a single INTOR design and for documenting the work. At each workshop session, decisions have been taken which are then incorporated in the national design contributions for the next session. These national design contributions are written following the outline of the final INTOR report. The final national design contributions will be published and a limited number of copies will be made available to each country for archival purposes.

Most of the design decisions for the INTOR Phase-I have now been taken by the end of session V. The self-consistency of these decisions will be confirmed at session VI (March 30 - April 10, 1981). A skeletal draft of the final INTOR report will be prepared at session VI, and responsibility for writing the various sections at home will be assigned. The draft report will then be reviewed, revised and approved at Session VII (June 22 - July 3 1981).

Special workshop groups have been organized to:

1. Assess the safety and environmental aspects of the INTOR design
2. prepare a cost/schedule/manpower assessment; and
3. define R and D programs to be recommended to the IFRC to resolve the most critical uncertainties that have been identified in the INTOR Phase-I design study.

# REPORT OF FOREIGN TRAVEL COST

1. Name of Traveler: Weston M. Stacey, Jr., Callaway Professor of Nuclear Engineering

2. Name of Contractor: Georgia Institute of Technology

3. Contract Number: DE-AS05-79ET-52049

4. Inclusive Dates of Travel: 1/19/81 - 2/6/81

5. Estimated Cost of Trip Shown on Form DOE-F-1512.1 ~~AEC-445~~: $3,765.00

6. Actual Cost of Trip Charged to AEC Funds: $3,032.28

E-26-669

# Final Report

Georgia Institute of Technology

Fusion Research Program

FY1981


prepared by

Weston M. Stacey, Jr.


April, 1981

Work was performed in the Georgia Tech Fusion Research Program during FY1981 under two research contracts with the Office of Fusion Energy of the U. S. Department of Energy: Contract No. DE-AS05-79ET-52049, "Perform Work in Support of the International Workshop on the Next Major Tokamak Experiment" and Contract No. DE-AS05-78ET-52025, "A Fusion Studies Program." Tasks under the former contract were related to coordinating the USA participation in the International Tokamak Reactor Workshop. Tasks under the latter contract were related to plasma systems analysis topics in support of the FED/ETF devices. A summary of the work performed, a listing of persons involved and a list of publications in which the work is described in detail are included in the attachment.

Attachment

Summary--Georgia Tech Fusion Research Program FY1981

A. INTOR (DOE # DE-AS05-79ET-52049)

1. Organized and technically managed USA INTOR conceptual design activity.

2. Performed studies in support of USA INTOR effort:

   a. Testing requirements and mission;
   b. Objectives and role in fusion program;
   c. Major parameter trade-offs.

3. Represented USA as steering committee member at INTOR Workshop sessions.

4. Coordinated documentation of INTOR work.

B. FUSION STUDIES (DOE # DE-AS05-78ET-52025)

1. Tokamak bundle divertor feasibility studies:

   a. Effect of field perturbation on magnetic islands in plasma (GTFR-15);
   b. Magnetic field line following code for bundle divertor design (GTFR-16);
   c. Bundle divertor conceptual designs (GTFR-20);
   d. "Hybrid" bundle divertor design concepts (GTFR-23).

2. Beam-driven impurity flow reversal in tokamaks:

   a. (Extension of theory to include temperature gradient effects. Not supported by this contract. [GTFR-14, GTFR-21]);
   b. Preliminary investigation of feasibility of flow reversal for impurity control in FED (GTFR to be published).

3. Burn control in tokamaks by magnetic field ripple:

   a. Development of a code package which incorporates ripple transport theory and an exact magnetic field representation to calculate ripple transport coefficients (GTFR-22);
   b. Preliminary investigation of feasibility of ripple for burn control in near-term tokamak reactors.

C. PERSONNEL INVOLVED

| Faculty | Graduate Students |
|---|---|
| R. G. Bateman | R. Bennett |
| J. N. Davidson | A. Engel |
| W. M. Stacey, Jr. | R. Morris |
| | P. Theriault |

D. PUBLICATIONS

1. INTOR

a. Stacey, W. M., Jr., C. A. Flanagan, G. L. Kulcinski, J. A. Schmidt, T. E. Shannon and 1980 U. S. INTOR Group, "U. S. Contribution to the International Tokamak Reactor Phase-1 Workshop," INTOR/80-1, Georgia Institute of Technology, Atlanta, GA, June, 1980.

b. Stacey, W. M., Jr., M. A. Abdou, J. A. Schmidt, T. E. Shannon and 1980 INTOR Group, "U. S. Contribution to the International Tokamak Reactor Phase-1 Workshop Conceptual Design," INTOR/81-1, Georgia Institute of Technology, Atlanta, GA (to be published, June, 1981).

c. Stacey, W. M., Jr., J. R. Gilleland, G. L. Kulcinski and P. H. Rutherford, "INTOR - A First-Generation Tokamak Experimental Reactor," GTFR-13 (2/80); also to be published in Nuclear Engineering & Design, Vol. 63, No. 2, North Holland Publ. Co., February, 1981.

d. Stacey, W. M., Jr., C. A. Flanagan, G. L. Kulcinski, J. A. Schmidt and T. E. Shannon, "Summary - USA Contribution to Session III Phase 1 INTOR Workshop," GTFR-17 (7/80).

e. Stacey, W. M., Jr., "The INTOR Workshop," GTFR-19 (10/80); also in Proc. ANS Topical Mtg. on Technology of Controlled Nuclear Fusion, October 14-17, 1980.

f. Stacey, W. M., Jr., M. A. Abdou, J. A. Schmidt and T. E. Shannon, "Summary of USA INTOR Conceptual Design Contribution," GTFR-24 (May, 1981).

2. FUSION STUDIES

a. Bateman, Glenn and R. N. Morris, "The Effect of Localized Magnetic Perturbation on Magnetic Islands in a Cylindrical Plasma," GTFR-15 (3/80); also to be published in Nuclear Fusion.

b. Morris, R. N. and Glenn Bateman, "DIVERT, A Divertor Magnetic Field Line Following Code," GTFR-16 (5/80).

c. Bateman, Glenn, R. N. Morris and P. Theriault, "Bundle Divertor Studies," GTFR-20 (10/80); also in Proc. ANS Topical Mtg. on Technology of Controlled Nuclear Fusion, October 14-17, 1980.

d. Bateman, Glenn and P. Theriault, "Hybrid Bundle Divertor Design," GTFR-23 (2/81).

e. Davidson, J. Narl and A. Engel, "Ion Heat Transport for Ripple Trapped Particles in Arbitrary Ripple Field Geometry," GTFR-22 (to be published).

f. Bennett, R. B. and W. M. Stacey, Jr., "A Preliminary Evaluation of Impurity Control by Neutral Beam Driven Flow Reversal in FED," GTFR-25 (to be published).

g. Bateman, Glenn, "Ripple Reduction Poloidal Field Coils for Tokamak," GTFR-26 (March, 1981).